NON-PUBLIC?: N
ACCESSION #: 9211170231
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Peach Bottom Atomic Power Station - PAGE: 1 OF 06
Unit 3

DOCKET NUMBER: 05000278

TITLE: An Automatic Scram During Instrument Testing and a Subsequent
Scram due to a Reactor High Pressure Condition and a Violation of
a Technical Specification Curve
EVENT DATE: 10/15/92 LER #: 92-008-00 REPORT DATE: 11/12/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(i) & 50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Albert A. Fulvio, Regulatory TELEPHONE: (717) 456-7014
Supervisor

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 10/15/92 at 2112 hours, a full Primary Containment Isolation System
(PCIS) Group I actuation occurred which caused a reactor scram. PCIS
Group II/III isolations occurred as expected. At 2316 hours, a second
reactor scram occurred on a reactor high pressure condition. This
occurred after the HPCI and RCIC systems tripped on a high reactor water
level condition which prevented them from being used for reactor pressure
control. The cause of the first scram has been determined to be an
unexpected actuation of very sensitive PCIS Group I isolation pressure
switches. The cause of the second scram has been determined to be less
than adequate command and control during the shift turnover process. The
cause of the pressure - temperature violation has been determined to be
stratification of reactor coolant in the bottom head region. After each

scram occurred, the appropriate PCIS and scram logics were reset and the affected systems were restored to the appropriate configuration. The events have been discussed with the involved individuals. Corrective actions will be implemented as appropriate for each event. No actual safety consequences occurred as a result of this event. Several previous similar events have been identified.

END OF ABSTRACT

Requirements of the Report

This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv) due to Engineered Safety Feature (ESF) actuations and to satisfy the requirements of 10 CFR 50.73 (a)(2)(i)(B) due to a Technical Specification (Tech Spec) violation.

Unit Conditions at Time of Event

Unit 3 was in the "RUN" mode at 100% of rated thermal reactor power. There were no systems, structures, or components that were inoperable that contributed to the event.

Description of the Event

On 10/15/92 at 2112 hours, an "A" channel half Primary Containment Isolation System (PCIS) (EIIS:JM) Group I isolation occurred after the performance of a Surveillance Test (ST) SI3P-1-14-C1CS "Calibration Check of Turbine First Stage Pressure Switch PS-3-05-14C". ST performance was immediately suspended until the cause of the half isolation could be investigated. While Operations personnel were scanning the PCIS relays to determine the cause of the "A" channel isolation, the "B" channel PCIS Group I actuated at 2116 hours. This resulted in a full PCIS Group I actuation which closed the Main Steam Isolation Valves (MSIVs). When the MSIV limit switches indicated that the valves were not full open, a reactor scram occurred. PCIS Group II/III isolations occurred as expected due to Reactor water level dropping below 0" as a result of void collapse upon insertion of the control rods. The High Pressure Coolant Injection (HPCI) (EIIS:BJ) system, Reactor Core Isolation Coolant (RCIC) (EIIS:BN) system, and the Alternate Rod insertion (ARI) initiated when Reactor water level dropped below the -48" set point to -50". Three Main Steam Relief Valves (MSRV) (EIIS:RV) lifted on high reactor pressure. The HPCI and RCIC were used in the Condensate Storage Tank (CST) (EIIS:TNK) to CST mode in conjunction with manual MSRV operation to control reactor water level and pressure. At 2125 hours, an Unusual

Event was declared in accordance with the Emergency Plan due to the HPCI and RCIC injection to the reactor from a valid initiation signal. The PCIS and the Reactor Protection System (RPS) (EIIS:JC) scram logics were reset by 2150 hours and the affected systems were restored to the appropriate configuration. The NRC was notified of the event via ENS at 2203 hours and the Unusual Event was terminated at 2300 hours.

At 2316 hours, a second reactor scram occurred on a reactor high pressure condition (1055 psig). This occurred after the HPCI and RCIC systems tripped on a high reactor water level condition (+45") which prevented them from being used for reactor pressure control. Following this scram, manual MSRV operation was used to maintain reactor

pressure at approximately 820 psig.
The NRC was notified of the event
via ENS at 0047 hours on 10/16/92. The RPS scram logic was later reset and the affected systems were restored to the appropriate configuration.

On 10/16/92 at approximately 1700 hours, during the review of ST-O-02B-500-3 "Recording of Reactor Vessel Temperatures", it was determined that a Pressure-Temperature violation, as specified by Tech Spec figure 3.6.2, occurred on 10/16/92 between approximately 0200 and 0945 hours. The reactor bottom head metal temperature was below that specified for the reactor pressure conditions.

Cause of the Event

The cause of the first scram has been determined to be an unexpected actuation of very sensitive PCIS Group I isolation pressure switches. Maintenance Instrument & Control (I&C) personnel (non-utility: non-Licensed) had performed an ST on a pressure switch located adjacent to the reactor low pressure instrumentation (PS-134s) which caused the PCIS Group I isolation. These devices are manufactured by Barksdale Control. The I&C technician, removing test equipment from the instrument in test and from the top of the instrument rack, inadvertently vibrated and actuated the PCIS Group I pressure switches. Following the event, several attempts successfully recreated the circumstances to prove this was the cause of the actuation.

The cause of the second scram has been determined to be less than adequate command and control during the shift turnover process. The Shift Supervisor (Utility:Licensed) (SSV) should have shifted the responsibility of vessel level control to the Reactor Operator (RO) and maintained the responsibility for pressure control with the operator manipulating the HPCI and RCIC systems. Had the level control been transferred to the RO, who was responsible for the Reactor Water Clean Up (RWCU) (EIIS:CE) system, which was in the vessel to vessel mode, the RWCU system could have been placed in the dump mode to control level. This transfer of responsibility was hampered due to the start of the on-coming shift turnover process. At this time, the oncoming shift had started the turnover process with the exception of the RO and the SSV. The SSV and the RO commenced reconstructing the event to write log entries prior to turnover. It was during this time period that reactor vessel level increased to the point where HPCI and RCIC had tripped leaving only the MSRV's for pressure control. The operator who was assigned pressure and level control attempted to lower level using RWCU in the dump mode, but, before level could be restored to the point where the HPCI and RCIC systems would be available, reactor pressure increased to approximately 1050 psig.

The cause of the pressure - temperature violation has been determined to be stratification of reactor coolant in the bottom head region. This resulted in a violation of the Tech Spec figure 3.6.2 pressure-temperature curves as indicated by reactor vessel skin temperature measurement near the bottom head region. The Recirculating (RECIRC) (EIIS:AD) pumps tripped due to a low reactor water level signal during the first scram. By the time that the operators were ready for restart of the RECIRC pumps, the bottom head drain temperature had dropped to less than that allowed by Tech Specs to restart a pump. The Recirculating pumps can not be started unless the coolant temperatures differential between the reactor dome and the bottom head drain is less than 145 degrees F. A contributing factor to this event was that no effective means existed to increase the flow rate out of the bottom head region to minimize stratification of reactor coolant. In addition, the second scram was not reset for more than six and one half hours which resulted in increased cold water flow into the reactor bottom head region via the Control Rod Drive (CRD) (EIIS:AA) system which increased stratification of reactor coolant. The operators were delayed in resetting the scram signals due to the high level of attention required to control reactor pressure and level utilizing HPCI, RCIC, and MSRVs. A contributing factor to this event was that the data tables in ST-O-02B-500-3 should have provided better human factors to ensure that data is verified to be within the Tech Spec pressure - temperature limits as specified in the ST text. The SSV failed to promptly review and evaluate the ST data sheets as specified in the ST text. This contributed to the violation not being recognized until after the event had occurred. Had the violation been identified during the event, actions may have been taken to depressurize the reactor to stay within limits. In addition, the scram may have been reset sooner to reduce the flow of cold water to the reactor bottom head region.

Analysis of Event

No actual safety consequences occurred as a result of this event.

All isolations and initiations functioned per design. Engineering has reviewed the pressure temperature violation and concluded that the thermal fatigue experienced by the Reactor Pressure Vessel during this event is bounded by previous analysis. Preliminary analysis has determined that the safety factor to brittle fracture, relative to reactor pressure, is at least 2.8. Further analysis is in progress to determine if the actual safety margin was within the guidelines specified in 10 CFR 50 Appendix G.

Corrective Actions

After each scram occurred, the appropriate PCIS and RPS scram logics were reset and the affected systems were restored to the appropriate configuration.

Station engineering has identified that the only Barksdale switches (PS-14A to 14D and PS-134A to 134D), which could impact plant reliability, are used on four instrument racks. Two racks on Unit 2 and two on Unit 3. As an interim corrective action, barriers have been placed in front of the Unit 3 instrument racks and will be installed on Unit 2 prior to Startup from the Refueling Outage. Warning labels have been installed on the instrument racks of concern for both units. The Unit 3 test taps for PS-14A to 14D have been modified to minimize the potential for inadvertent actuations during future testing and the Unit 2 test taps will be modified prior to startup. Engineering has been requested to evaluate the possibility of replacing the existing instrumentation with a type which is less susceptible to inadvertent actuations. The first scram has been discussed with the involved individuals. An I&C training bulletin has been distributed to all I&C personnel to notify them of the potential risk.

The second scram has been discussed with the involved individuals. Team building training will be performed for the appropriate Operation personnel to address lessons learned from the second scram. This will include turnovers after transients with emphasis on command and control in conjunction with role clarity.

The pressure - temperature violation has been discussed with the involved individuals. The data tables in ST-O-02B-500-2(3) have been enhanced to include a reference to the applicable Tech Spec figure 3.6.2. In addition, engineering has been reviewing the need for a motor operated valve on the RWCU suction line which could be throttled to increase the bottom head drain line flow rate. The increased flow through the bottom head drain line would minimize reactor coolant stratification. This would allow the RECIRC pumps to be promptly restarted following future similar events. In addition, training will be provided to the applicable Operations personnel to increase their sensitivity in monitoring and controlling reactor pressure and temperatures during transients to ensure that mitigating actions are taken to limit the duration and severity of future similar events. in addition, another similar event on Unit 2 is currently under investigation. Any additional causes and associated corrective actions will be submitted under LER 2-92-024.

Previous Similar Events

One previous similar event (LER-3-85-022) has been identified which involved a Reactor Protection System actuation when an instrument was inadvertently bumped. Since this event involved the Scram Discharge Volume high water level scram switches, which are different from the Barksdale pressure switches, it is not expected that any corrective action from the previous event would have prevented this event. The corrective action taken has part of this event are expected to reduce the potential of future inadvertent actuations.

Four previous similar events (LER-2-85-05, 2-85-16, 3-89-05, and 3-89-06) have been identified which involved a Reactor Protection System high pressure scram actuations. Since these events involved a leaking instrument valve, and less than adequate planning and blocking, which is different from the failure to monitor reactor pressure, it is not expected that these corrective actions from the previous event would have prevented this event. The corrective actions taken has part of this event are expected to minimize future similar actuations.

No previous similar events have been identified which involve pressure - temperature violations. However, three previous similar events have been identified which involved heatup / cooldown violations or the failure to record these parameters due to a less than adequate surveillance test. The corrective actions implemented as part of these events addressed enhancements in recording data. However, they did not address acting on the data. Therefore, it is not expected that the corrective actions taken as part of the previous events would have prevented this event.

The corrective actions being taken has part of this event are expected to minimize future similar violations.

ATTACHMENT 1 TO 9211170231 PAGE 1 OF 1

CCN 92-14130

PHILADELPHIA ELECTRIC COMPANY

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KEN POWERS
PLANT MANAGER November 12, 1992

Docket No. 50-278

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Licensee Event Report
Peach Bottom Atomic Power Station - Unit 3

This LER concerns an automatic scram during instrument testing. Subsequently, a second scram due to a reactor high pressure condition and a violation of a Technical Specification pressure - temperature curve occurred.

Reference: Docket No. 50-278
Report Number: 3-92-008
Revision Number: 00
Event Date: 10/15/92
Report Date: 11/12/92
Facility:
Peach Bottom Atomic Power Station
RD1, Box 208, Delta, PA 17314

This LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv) and 10 CFR 50.73(a)(2)(i).

Sincerely,

cc: J. J. Lyash, US NRC Senior Resident Inspector
T. T. Martin, US NRC, Region I

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